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To: Dr. D. Mills
Director of Research

CORROSION AND TRITIUM TECHNOLOGY SECTION
1989 REVIEW AND 1990 WORK PROGRAM

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The key projects and achievements of the Corrosion and Tritium Technology Section (C&TT) during 1989 are reviewed and the work program for 1990 is highlighted. The section activities continue to be concentrated largely in the areas of Zr alloy corrosion/hydrogen ingress mechanisms, steam generators/heat exchanger corrosion mechanisms, tritium handling/storage processes, and tritium interactions with metals.

INTRODUCTION

This report presents an overview of the C&TT Section work programs, the key progress achieved during 1989, and the direction in which the work program will be focused in 1990.

The mission of the section is to carry out R&D, perform technical investigations and act as a corporate centre for expertise in the areas of metallic corrosion mechanisms/control, and tritium-metal interactions/tritium handling. The section is accordingly divided into two main work groups - Corrosion (60%) and Tritium Technology (40%).

CORROSION

Pressure tube Zr alloy corrosion and hydrogen ingress mechanisms work remains high priority accounting for 25% of the section's efforts. The critical need is to understand these processes so that reliable predictions of current pressure tube life can be made, and also so that improved materials and/or operating procedures can be established to extend the life of these components. Research is concentrated on heat transport system coolant effects, alloy microchemical/microstructural effects, and annulus gas effects.

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Nuclear Steam Generator tubing corrosion mechanism research also continues to be a key activity and has been gradually increasing, now accounting for 17% of the section's efforts. Despite an overall very good record of performance reliability, there are increasing concerns as to the long term integrity of alloy 600 tubing. Anticipatory research is being conducted in the areas of stress corrosion cracking, corrosion fatigue, and under deposit localized corrosion degradation. Generic degradation of this kind had led to large incapability in nuclear steam generating equipment around the world. Deposit build-up in Bruce 'A' generators is currently causing deratings due to flow blockage at tube support plates. Under these conditions, there is an increasing risk of corrosion related degradation to the tubing.

As a result of failures in a number of nuclear plant heat exchangers, a substantial research effort continues to quantify under-deposit pitting and other local corrosion degradation mechanisms. This work accounts for roughly 10% of section activity. Work is focused on establishing the role of alloy chemistry and microchemistry, microbiology, and deposit formation. Electrochemical techniques and high sensitivity surface analysis methods combined with laboratory and site exposure test rigs, form the basis of these activities.

The remaining 8% of activities in the corrosion program involve corrosion engineering support to nuclear, thermal, and hydraulic plant equipment, as well as transmission, distribution and other related equipment. The main efforts here are directed toward corrosion failure evaluation, corrosion control methods, corrosion monitoring, and selection/evaluation of materials to reduce or eliminate corrosion. In the last year these activities have increased. This application of expert corrosion advice and testing provides large cost benefit ratios, often exceeding a factor of 10 or more. During 1990 even more activity is expected to address possible corrosion implications of Zebra Mussel infestations in the Great Lakes, the anticipation of increased S/G tubing surveillance examinations and the necessary development of electrochemical monitoring technology.

TRITIUM TECHNOLOGY

The Tritium Technology (TT) group was formed in 1987 by the amalgamation of two separate work groups within the Section. Engineers responsible for the design and construction of the Tritium Immobilization System (TIS) for the Darlington NGS Tritium Removal Facility (TRF) were combined with the scientists responsible for tritium research in support of fission and fusion applications.

A main continuing component of this work involves metal getter research and development, which accounts for 15% of the section work activity. Getters are an extremely safe and effective way of storing and immobilizing large volumes of hydrogen isotopes. This work directly supports Darlington TRF. Activities involve characterization/quantification of getter properties and kinetics and the development of improved getter alloys. This work has led to the design

and development of practical and efficient storage and recovery devices/systems which are being commercially marketed by the Canadian Fusion Fuels Technology Program (CFFTP).

The study of tritium interactions with materials, accounts for another 10% of section effort. Hydrogen isotopes due to their small atomic volume and mass, are easily adsorbed/absorbed and rapidly permeate most engineering materials. Add to this the emission of an energetic beta particle (resulting from the decay of the tritium atom) and there exists the potential for progressive material property degradation. Understanding these processes is crucial to understanding tritium effects in fission and potential fusion material processes. Both applied and basic work continues to investigate the nature and consequences of material degradation, unique gas separation/filtering methods, decontamination, and advanced "Tritium Light" concepts/devices.

In order to study tritium materials interactions, gas handling systems/processes are required. The development of these facilities accounts for the final 15% of the section work program. A divisional tritium laboratory is in operation. Operation/development of this lab facility is providing invaluable knowledge/experience in the development and application of tritium monitoring, tritium control and cleanup of work environments, specific tritium handling practice and tritium materials compatibility. Support in commissioning and operating the tritium immobilization system of the Darlington Tritium Removal Plant and the design/construction of a tritium dispensing loop (TAP) is a direct application of the tritium lab operational experience.

FUNDING/CLIENTS/STAFFING

In 1989, the 2.7M$ work program of the C&TT section was roughly 80% transfer funded. By far the bulk of the work was in the Nuclear Generation program area with principal clients being Technical and Training Services Division, Nuclear Generation Division, and Design and Development Division. Work performed on behalf of the CANDU Owners Group (COG), accounted for 90% of all corrosion related transfer funds and for 1990 this will continue. This large proportion of committed work continues to have a major impact in terms of our ability to respond to short term corrosion work requests from other clients. The current volume of requested corrosion R&D work is fully at the limit of available staff resources to undertake the work. If anticipated increases in 1990 in corrosion work to support S/G tube surveillance and Zebra Mussel programs are realized, we will be forced to delay or drop some committed priority work. Transfer funding of the Tritium Technology program accounts for about 60% of the total program costs. Four main client groups, the Canadian Fusion Fuels Technology Program (CFFTP), New Business Ventures Division (NBV), the Darlington Project, and the National Fusion Program, share the total effort.

The staff of the Corrosion and Tritium Technology Section are listed in Table 1. During 1989, two additional researchers were temporarily attached to the Tritium Technology Group: one from CFFTP, and a University of Toronto

Masters Student. Internal reports and external publications/presentations, visits and visitors are also attached. The following sections provide highlights of the key activities of the work program in terms of significant 1989 progress and planned objectives for 1990.

CORROSION PROGRAM

Zirconium Corrosion

Deuterium concentrations in the body of CANDU Zr-2.5 wt% Nb pressure tubes increases with operating time. Removed an/or scraped pressure tubes from Pickering and Bruce reactors have shown highly variable levels of deuterium with a few tubes containing levels of sufficient magnitude to be of concern. The oxide acts as a diffusional barrier preventing significant deuterium ingress. Deuterium may be picked up from the reactor coolant as a result of corrosion reactions and from the annulus gas system.

Out-reactor long-term high temperature (250°C, 310°C and 360°C) aqueous exposures are being carried out with samples of Zr-2.5 wt% Nb and Zircaloy-2 taken from unirradiated pressure tubes, their offcuts and hardness rings, with a range of impurity contents, thermomechanical histories and surface conditions. Results are given in OHRD Report 89-167-P. To date exposure durations of over 600 days have been useful in distinguishing between the short-term transition (pre-breakaway) performance of these materials (Figure 1). Differences in corrosion and deuterium uptake rates between samples have been quite small. Some evidence has been found that oxidation performance improves with increasing Fe content. Longer term exposures are required to substantiate these differences and provide information on the relationship between acceleration in corrosion rates and sample condition. This work will continue throughout 1990.

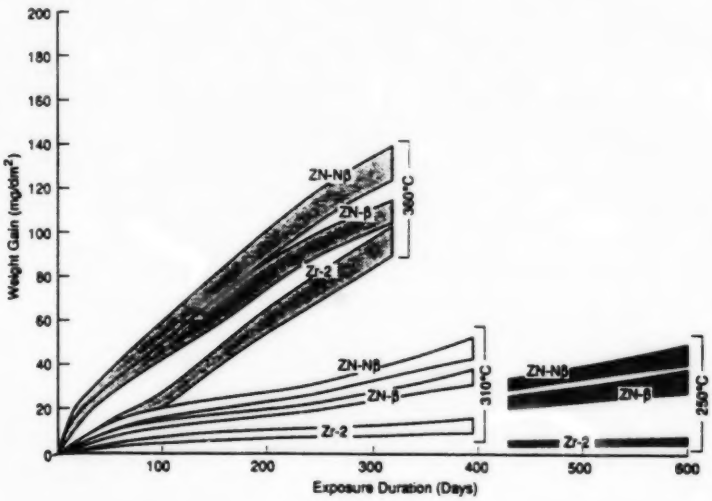


Figure 1

OUT-REACTOR CORROSION KINETICS FOR Zr-2.5 wt% Nb β-QUENCHED (ZN-β) AND NON β-QUENCHED (ZN-Nβ) AND ZIRCALOY-2 (Zr-2)

Out-reactor gaseous exposures are in progress. Their objective is to continue to investigate reaction mechanisms of deuterium/oxygen with Zr-2.5 wt% Nb in gaseous environments with emphasis given to the development of kinetic data (reaction rates) as a function of temperature, gas environment and material surface condition. More precise reaction rate data will also be obtained for deuterium on clean Zr-2.5 wt% Nb surfaces at CANDU operating temperatures, and oxide breakdown as a function of time and temperature will be investigated. During 1989 standard ultra-high vacuum (UHV) stainless steel components were used in completing the building of a testing chamber connected to a turbomolecular pump station. The chamber total pressure is monitored by a BALZERS IKR 020 inverted magnetron gauge with TPG 300 controller. A BALZERS QMG 112 A mass spectrometer is also attached to the chamber as a residual gas analyzer. Resistive heating bands with electronic controls are used to maintain the chamber temperature within 1/2°C. A gas-dosing system capable of dosing a known amount of gas molecules was also built out of UHV standard components. A schematic diagram of the setup is shown in Figure 2. Significant experimental results are expected in 1990.

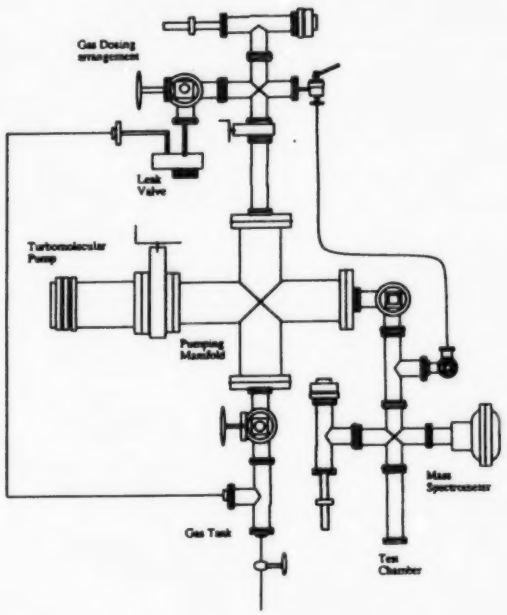


Figure 2
A schematic diagram of the experimental setup

The characterization of near-surface chemical and microstructural changes occurring in pressure tubes following in- and out-reactor exposures is also continuing. Thin rectangular samples have been cut from inside and outside

surfaces of removed pressure tubes at selected locations corresponding to differing hydrogen uptake rates and reactor environments. Samples have also been prepared from coupons from new pressure tubes following out-reactor exposures in elevated temperature aqueous/steam environments. SEM examination of oxide undersides was performed by etching away the metal. Since the oxide structure observed by SEM appears to be independent of etching time, this surface probably represents the oxide growth front reasonably well. For TEM examination of oxides near the metal-oxide interface, samples are glued together with epoxy, cut into rods, encased in epoxy, and foils are prepared from these sections by dishing and ion milling techniques. Secondary Ion Mass Spectroscopy (SIMS) is performed to determine depth profiles of various oxide and metal chemical species including hydrogen. Ridges have been observed on oxide undersides corresponding to the location of the β-Zr phase in Zr-2.5 wt% Nb, which indicates locally higher corrosion rates of this phase. Holes have been observed on both inside and outside surface oxide undersides in samples taken from the location of peak hydrogen content in P3L9 at 5.9 m from the inlet (see Figure 3). These holes are formed when metallic protrusions into the oxide are etched away during sample preparation. The unique interfacial structure of outside surface oxides at this location, with their basketweave appearance, seems to result from higher local rates of oxidation, or lower oxide dissolution rates, at the intersections of the troughs of the grinding marks and the β-Zr phase.

This work will be extended in 1990 by continuing to apply quantitative microstructural and microchemical analysis (TEM/SIMS) to additional samples from pressure tubes removed from reactors and from controlled out-reactor tests. Correlation of results will be studied using physical models to help establish the key controlling mechanistic features describing ingress of deuterium.

Figure 3 - Structure of Oxide-Metal Interface on Outside Surface of P3L9 at 5.9 m from the Inlet

A supporting computer modelling program is underway to further analyze data to understand the influence of variables and predict corrosion and deuterium uptake in operating reactor pressure tubes. Additional data from removed pressure tubes P3L9, B1L8, P4D19, and Pickering 3 and 4 and Bruce 2 scrapings were included in the data base. The data base includes all data on hydrogen/deuterium contents and oxide thicknesses on scraped and removed pressure tubes. It is now available in LOTUS, SAS and DBASE formats. Work continues to develop an explanation for development of peaks in deuterium concentration profiles at pressure tube outlet (and possibly inlet) regions. Peaks in axial deuterium concentration profiles have now been observed in outlet regions of P3D19 and B2L8 (and B2S4) in addition to P3L9 and P4K10 (and P1 and P2 Zircaloy-2 pressure tubes). These peaks appear to correlate in general with the shape of the flux profile. A mathematical model has been constructed in conjunction with T. Byrne and D. Metzger (OHRD Applied Mechanics) to determine the deuterium ingress necessary to result in measured concentration distributions found in pressure tubes. The model accounts for atomic diffusion in calculating how much deuterium is picked up at each point and how much arrives via diffusion from neighboring regions. The model is used to analyze data from the removed pressure tubes P3L9 and P4D19. While limited available concentration data hinder the ability to reach definite conclusions, both possible and improbable ingress scenarios can be identified. The model can account for spatial (temperature) and time variation of the deuterium uptake rate, and solve for 1D or 2D diffusion. Deuterium uptake rates and total amount of deuterium uptake at any location may be determined, given the measured deuterium concentration profile. Axial deuterium concentration profiles vs time have been generated for P3L9, assuming various deuterium source scenarios. Maintenance, addition to, and further analytical development of computer mathematical modelling concepts will continue in 1990.

Corrosion Fatigue and Steam Generator Tubing Alloys

Corrosion fatigue cracking has not been an extensive problem for alloy N06600 steam generator tubing; however, the incidence of documented corrosion fatigue failures is rising. Failures due to this mechanism have occurred at:

> Oconee, a B&W once-through unit operated by Duke Power Co.,
> Ontario Hydro's Bruce NGS 'A',

and more recently at

> North Anna, a Westinghouse unit operated by Virginia Power Co.,
> and Indian Point 3, a Westinghouse unit operated by the Power
> Authority of the State of New York.

The extent of environmental contribution in these failures is unknown. There are no known incidences of corrosion fatigue cracking of N08800 in steam generator service.

Since a synergistic stress-corrosion/corrosion-fatigue mechanism was believed responsible for the Bruce NGS failures, a project was initiated in 1988 to investigate the high cycle corrosion fatigue behaviour of N06600 and N08800 tubing of CANDU manufacture under conditions relevant to CANDU steam generator operation.

The fatigue response of CANDU N06600 and N08800 tubing is being determined over a range of loading and chemistry conditions encompassing those occurring in service. In order to make efficient use of limited resources, an orthogonal experimental design is employed.

High cycle fatigue is often simulated in laboratory investigations by testing smooth or notched specimens under cyclic loads. In this program, a U-bend tubing specimen geometry was chosen because it offers advantages in containing an environment, as well as in the loads required to produce high stress levels, and the ability to accommodate a mean stress.

The development of the test apparatus has been completed. In 1989, design changes were made to the sample grips, and linearization of the load time has been improved by the incorporation of a cross-roller guide. Pressure, rather than load is now used as a feedback control method to stop the test if failure occurs before the set number of cycles. Four testing systems, each operating independently, are now available.

A number of test series were completed in 1989. Figure 4 presents a typical fatigue staircase resulting from a test series, and Table 1 summarizes the analysis of the staircase data for the various series.

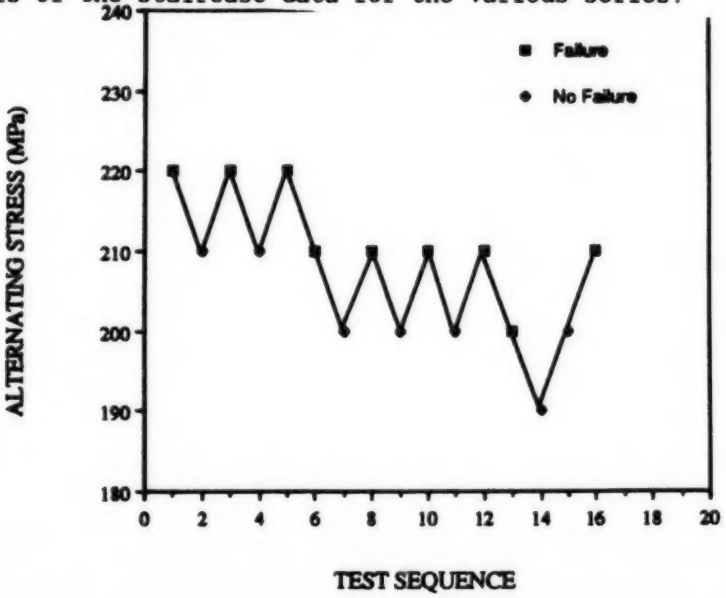


Figure 4 - Staircase fatigue test at $10^{6.5}$ cycles for alloy N06600
Test Condition: as bent, open ended tube, laboratory air, test rig B, mean stress 240 MPa.

TABLE 1

RESULTS OF FATIGUE TESTS

Test Series No.	No. of Runs	Test Rig	Tubing Material	Condition	Test Temp.	Environment	Applied Mean Stress (MPa)	Fatigue Strength (MPa) at $10^{6.5}$ cycles	
								Mean	Standard Deviation
SN1	10	A		as bent	RT*	air	200	132.5	10.5
SN2	16	B		as bent	RT	air	240	196.5	5.5
SN3	10	B	N06600	as bent pressurized	RT	air	240	180	18
SN4	2	B		as bent, peened pressurized	RT	air	240	ip	--
SN5	3	B		as bent, peened	320°C	10% NaOH	240	ip	--
SN6	5	A		as bent	RT	air	200	130	8
SN7	7	A	N08800	stress relieved	RT	air	200	160	8
SN8	3	B		as bent	RT	air	200	nf	--

RT - room temperature (~21°C)
ip - in progress
nf - no failure
Test Rig: A - first design
B - second design equipped with cross roller guide

In the as-bent condition, specimen geometry and stressing method result in crack nucleation on the outside surface of test samples. Preliminary tests with peened samples have shown an inside surface crack nucleation region. Inside surface initiation is desirable to obtain an environmental influence with the solution on the inside of the tube.

Both bending and peening of the tubes produce residual stresses that will affect the fatigue behaviour of the alloys. Although this will not hinder relative assessments of corrosion fatigue, knowledge of these stresses is useful. Thus, detailed stress analyses of the tube samples (as-bent and peened) are in progress to quantify the stress distribution in the samples.

Testing of alloy N06600 in laboratory air will be completed in 1990. The scoping test matrix for environmental effects is a seven factor, two level orthogonal design. The input factors will include heat-treatment, frequency, mean stress, pH, anionic constituents, and number of cycles. Eight test runs, each involving a fatigue staircase of about twelve samples, are required to complete the orthogonal test matrix. It is expected that up to six test runs will be completed in 1990.

Testing of peened alloy N08800 in air will be initiated in 1990.

Stress Corrosion Cracking of Steam Generator Tubing Alloys

Stress corrosion cracking of N06600 (alloy 600) tubing has proven to be a major reliability problem in PWR steam generators. Based on this and extrapolation of available research data, it is estimated that cracking of CANDU steam generator material could begin to become a problem after 15 years of operation. Although, only one operational failure of N08800 (alloy 800) has been reported, cracking resistance of this alloy may be less than that of N06600 in certain environments, particularly very strong caustics.

This project was initiated to determine the low temperature stress corrosion cracking resistance of CANDU steam generator tubing alloys N06600 and N08800. To do this, samples of steam generator tubing with simulated process heat treatments are stressed and exposed at steam generator temperatures. Aggressive (high pH) environments are employed to produce results in a reasonable time. The main variable in testing is temperature.

Heat treatment and initial metallographic assessment of CANDU N06600 tubing was completed in 1987. Several autoclave tests in 350°C caustic solutions were completed in 1988, with durations of up to 2000 hours. Included in these exposures were various heat treatments of CANDU N06600, two lots of PWR tubing received from EPRI (high and low susceptibility heat treatments), and N08800.

In 1989, a total of over two hundred N06600 and N08800 C-rings were tested in autoclave exposures. These tests, with durations of up to 4000 hours, were carried out in 10% NaOH at 350°C and 290°C. Methods for crack measurement and data analysis were improved in 1989. The sample evaluation method now

involves cutting multiple cross-sections from each C-ring and measuring crack depths on these. The maximum crack length for a C-ring is then taken as the absolute maximum of the measurements on the individual cross-sections.

Table 2 summarizes the maximum crack length data for testing at 350°C, and gives relative cracking resistances for the various heat treatment conditions. The trends in maximum crack length for alloy N06600 are well behaved with the exception of the CANDU 607°C heat-treated tubing at 2000 hours exposure. In this case the relative cracking resistance is considerably lower than expected, and is not consistent with the improvements obtained by the lower temperature heat treatments. Some microstructural non-uniformity has been observed in heat-treated tubing; this is being examined further as a possible reason for the observed behaviour. Alloy N08800 tubing suffered only minimal cracking in these tests. The observed cracks resulted from locally high stresses associated with surface defects; the applied stresses were insufficient to cause cracking.

TABLE 2

SUMMARY OF C-RING RELATIVE CRACKING RESISTANCES IN 10% NaOH AT 350°C

Alloy	*Sample Condition*	*Cracking Resistance Relative to CANDU N06600*[+] *2000 hr*	*4000 hr*
N06600	CANDU	1	1
	+500°C/10 hr (BAU)	2.5	1.5
	+550°C/10 hr (BAT)	5	2
	+607°C/10 hr (BBF)	2	13
	EPRI-Lot 2 (MAL)	0.1	---[++]
	EPRI-Lot 1a (MAH)	1	3
N08800	CANDU	>40	>40
	+607°C/10 hr (BBF)	>40	>40

[+] The crack length ratios are based on a log-normal distribution of maximum crack lengths. All applied stress levels are combined for each alloy condition.

[++] Calculation not possible because cracks typically extended the full thickness of the C-rings.

BAU	Bruce A U-bend	TT	Thermally Treated
BAT	Bruce A Tubesheet	MAH	Mill Annealed, High Temperature
BBF	Bruce B Full-Vessel	MAL	Mill Annealed, Low Temperature

Examination of samples after 3000 hours exposure at 290°C revealed minimal cracking. Data is required at longer exposure times in order to estimate crack growth rates and apparent activation energies. However, cracks are shorter than expected and this suggests a very strong temperature effect.

Microstructural studies employing conventional metallography and analytical high-voltage electron microscopy are underway. The electron microscopy has generally confirmed the structures observed by light microscopy, and is providing additional detailed information on grain boundary carbi[illegible] distribution. Grain boundary segregation is also being investigated with the analytical high-voltage electron microscope.

Problems encountered with specimen fabrication caused a delay in testing at 290°C and 260°C. A new fabrication route, using EDM (electric discharge machining) has been evaluated and adopted for future use. A stock of EDM-fabricated C-rings has been procured and heat-treated.

In 1990, autoclave testing will continue at 290°C and 260°C in 10% NaOH. Based on the early results at 290°C, inordinately long exposure times may be required to induce cracking at 260°C. Therefore, alternative methods will be considered to obtain the desired information in a shorter time. Analytical microscopy will continue to obtain the desired information in a shorter time. Analytical electron microscopy will continue to provide microstructural and microchemical information.

Heat Exchanger Corrosion

Ontario Hydro's nuclear plants have experienced several heat exchanger (HX) failures due to pitting on the lake water side of the tubes. Although various alloys and HX 's of various configurations have suffered from this problem all failures have been the result of crevice corrosion beneath adherent sand, clay and carbonate mineral deposits. Alloy NO8800 moderator HX tubes have been particularly prone to leakage as a result of underdeposit corrosion. In addition the alloys (NO8825 and NO8028) which have been used to replace alloy NO8800 are also exhibiting pitting corrosion after only a short period of service.

To investigate ways of eliminating corrosion of moderator heat exchangers a number of studies have been initiated which include field and laboratory experiments. Electrochemical, immersion experiments and microbial studies are included in the laboratory investigations.

The field studies are directed towards the generation of corrosion information under simulated service conditions. Two model heat exchangers are currently in operation and are situated at Pickering and Darlington NGS. Analysis of tubes removed from these rigs indicate that fouling is initiated by the formation of a biofilm, which consists of an extracellular polymeric material (EPM) and bacteria. The EPM acts as a glue, sediment particles which settle on the tube surface become immobilized. These particles become substrates for the nucleation of calcareous material. The final and ongoing stages are

probably a combination of the above. Eventually the surface becomes covered in a thick scale which prevents access of oxygen to the surface to repair defects that are formed in the normally protective film covering the metal; the result is pit initiation.

Alloy NO8800 and NO8028 tubes exposed to the simulated HX environment have exhibited pitting corrosion after exposure times of one and two years respectively. As the rig appears to be faithfully simulating the conditions in a full size heat exchanger, based on service data for alloy NO8800 and rig data for this material, it appears that replacement alloy (NO8028) may have a shorter service life than intended.

In 1989 most of the effort on the HX program was expended on the microbiological aspects of heat exchanger corrosion. The development of a method of testing the susceptibility of HX materials to Microbiological induced corrosion (MIC) has been the main area of focus. Corrosion caused by anaerobic sulphate reducing bacteria (SRB) was the field of interest as the corrosion products found on HX tubes indicates that they may be implicated in the corrosion problem.

A collaborative effort between the Biology, Process Chemistry, and Corrosion and Tritium Technology Sections has resulted in a much better understanding of the factors which must be carefully controlled when attempting to perform experiments which involved bacteria.

Preliminary experiments involving the exposure of alloy NO8800 to culture medium inoculated with sulphate reducing bacteria in a batch culture indicated that pitting in this environment was possible. After an exposure period of sixteen weeks pits were observed in alloy NO8800. The morpholoy of the pits strongly resembled those found in heat exchangers. The results of this study suggest that bacterial coverage is an important factor in the corrosion process.

Large scale corrosion test rigs were fabricated and commissioned. The rigs work on a semi-continuous circulating culture medium principle. Testing, including evaluating different culture media, was performed. Results obtained for carbon steel indicated a high level of hydrogen sulphide generation. Hydrogen sulphide results from SRB activity. The carbon steel corroded very rapidly in less than one week. Examination of the surface of the coupons with laser ionization mass analysis (LIMA) revealed the presence of iron sulphide corrosion product on the surface. The sulphur isotope ratio clearly indicated that it had been biogenically produced. This was evidence that the rig was capable of sustaining a viable SRB culture. Hydrogen sulphide monitoring and bacterial population counts also indicated that an active bacterial culture could be maintained in the rigs.

Testing of NO8800 coupons in the rigs however, did not lead to corrosion. This inspite of the high population of bacteria present in the rig, which was likely much higher than in the batch cultures. Examination of the surface of the coupons with the scanning electron microscope after exposure revealed very

few bacteria attached to the surface. It appears from the batch culture studies and the literature that bacterial attachment is a very important factor with respect to the corrosion of passivating materials. The result was not surprising. It should be noted that attachment is not as important with carbon steel. Three obvious reasons for this behavior are; i) the presence of shear stresses at the metal surface due to flow, research elsewhere has indicated that this is a major consideration, ii) the bacteria prefer to be in the planktonic form in nutrient rich culture medium and iii) the orientation of the coupons in the rig.

Further studies aimed at determining what can be done to improve attachment, and thus accelerate attack, are underway. Coupons preconditioned with agar containing culture medium seems to be one approach that will promote attachment.

A study was also initiated which was aimed at establishing the importance of the various additions in culture media. The objective of this work was to formulate a culture medium which could be used for accelerated testing of heat exchanger materials.

The development process for silane coating nickel alloys was successfully accomplished (part of the Directors Fund program). The Process Chemistry Section formulated a silane which when attach to the tube had non stick qualities and would prevent fouling build-up. Numerous test parameters were evaluated using an orthogonal experiment design. The effect of varying the important process factors was established by surface analysis (XPS) and contact angle. A process was established which produced the required coating.

Work Planned in the HX area for 1990 includes the examination of a bundle removed from the Pickering and Darlington rigs after two years exposure. Attempts will be made to microsection undisturbed pits ie, pits which are covered in deposit. The aim of this work will be to further understand the sequence of events which leads up to loss of passivity and pitting. It is hoped that this work will provide additional mechanistic data. Work on the development of an accelerated test will be completed. Sample examination which includes determining pit and bacteria populations, has yet to be performed. This data will be related to culture medium composition. The final part of the test will involve electrochemical monitoring. Laboratory and field studies aimed at the development of electrochemical noise techniques is also planned. The CAPCIS equipment will be used to study the effectiveness of silane coatings on HX tubes. This will be done on the PNGS HX rig. The object of the laboratory studies will be to establish whether this tool can be used to distinguish between microbial corrosion and chloride attack.

A new program area involving examining the effects of zebra mussels on heat exchanger materials will also be initiated. The effects of chlorination and ozonation (both of these biocides are candidates for controlling zebra mussels), will be investigated with respect to their long term effect on power plant material degradation. The electrochemical noise technique will be used

to monitor the effects of these biocides on corrosion. Cathodic protection will also be investigated as a potential method of reducing accelerated corrosion that might be produced by zebra mussels attachment.

Corrosion Engineering

As always seems to be the case, work in this area was very diverse and mostly reactive.

The main areas of activity during 1989 were; i) the Lennox cathodic protection system redesign, ii) manure irrigation corrosion studies, iii) hydraulic generating station service waterpipe replacement study and iv) the Bruce heavy water plant "D" corrosion survey. Water heater work was less prominent in 1989 due to changes in regions and areas. The cathodic protection system design was completed and has been installed. Problems were encountered with the manure irrigation study. Manure irrigation was performed in late fall during a very wet period. It is feared that the rain very quickly washed the manure from the corrosion monitors and as a consequence had very little effect on corrosion. Samples of pipe from hydraulic plant were examined for corrosion. Corrosion rates were obtained and reported. The Bruce Heavy Water Plant "D" corrosion survey was completed. General corrosion at a low level appears to be continuing in the mothballed heavy water plant.

The main activity for 1990 is the Pickering vacuum building corrosion inspection. It is also anticipated that the water heater work will once again gradually increase in importance and that the manure irrigation study will continue.

Nuclear Waste Container Corrosion

The nuclear waste container corrosion program is aimed at determining the corrosion performance of candidate materials for the fabrication of containers for the future disposal of used nuclear fuels in hard rock formations. Copper, because of its relative inertness to many chemicals, is being considered as a potential container material. The copper corrosion program is designed to provide information for estimating a corrosion allowance thickness that would meet the minimum five hundred years lifetime specification of the container.

In 1989 major findings of the work were:

1) Copper pits and corrodes non-uniformly giving a generally roughened appearance in a projected saline ground-bentonite vault environment.

2) Oxidant content has a significant accelerating influence on the corrosion rate of copper, and salinity at all concentrations has no apparent effect.

3) Irradiation has a generally beneficial effect on the corrosion of copper. The mechanism behind this phenomenon has not been studied and is as yet unknown.

4) The porosity of the corrosion product has a great influence on copper's corrosion rate and the morphology of attack. A more porous product produces a higher corrosion rate and a rougher surface with a higher density of pits.

5) A general corrosion allowance of 2 mm should be sufficient in meeting the lifetime specification of the container in a sulphide free environment at 302K with an oxygen concentration of less than 1 mg/L. If the environment is equilibrated with an atmosphere of air, a 5 cm allowance is estimated. The detailed effect of pitting has not been considered in these estimates.

The 1990 program will be directed towards studying the effect of pitting on copper's corrosion performance. Two investigations are in progress:

1) Long term immersion testing, this involves exposing copper for up to five years in saline ground water.

2) Pitting current measurements.

TRITIUM TECHNOLOGY PROGRAM

OVERVIEW

The Tritium Technology group is dedicated to the study of tritium and its interaction with materials including the daily management of the Tritium Laboratory.

Two facilities are committed to this area of study: a Hydrogen Laboratory and a companion Tritium Laboratory. Proof of principle tests are pursued in the Hydrogen Laboratory and pave the way for authentication in the Tritium Laboratory in a cost effective and rapid manner usually with trace quantities of tritium. Several experimental stations have developed over the past few years to address the varied needs of the field. The Hydrogen Laboratory contains: the Bulk Getter Test Stand, to evaluate materials for the storage of tritium, the Scavenger Test Stand, to evaluate the ability of materials to scrub tritium from inert gas streams at the parts per billion level and a Permeation Facility, to investigate the hydrogen permeability of metals. Currently, the latter facility is investigating the feasibility of separating hydrogen isotopes from the exhaust of a tokamak by means of permeation. The Tritium Laboratory houses three stations which are used by Tritium Technology: the Component Test Stand for the evaluation of components for tritium service, the Thin Film Facility for the development of solid matrix tritium lights and the Decontamination Center for the study of detritiating metal surfaces.

Tritium Technology has concentrated on the management of tritium gas. This endeavour has manifested itself through developments in four distinct areas:

permeation, interaction of tritium with hydride forming materials, development of tritium compatible components and systems and the operation of the Divisional Tritium Laboratory. From this perspective, the group responds to needs of several clients: the Tritium Removal Facility at Darlington NGS, New Business Ventures, Canadian Fusion Fuels Technology Projects, and the Nuclear Engineering Department. The developments in each area during 1989 are discussed in separate sections below.

Tritium Storage Beds

Uranium offers several advantages as a storage medium. It can store tritium within its metal lattice at densities two times that of liquid tritium. Recovery of tritium is effected by heating the uranium tritide to 400°C. Under accidental exposure to air a negligible fraction of the stored gas will be released to the environment. The design of reliable storage beds requires an understanding of the heat transport from the reaction zone and to the environment during tritiding and in the control of uranium fines production. To this end a model which accounts for the thermal inertia of the primary vessel, the gas conductance of the filters used to contain the uranium fines within the primary vessel and the reaction kinetics of the gas solid reaction has been developed and tested. This model can guide the design of a storage bed to ensure its optimal performance. Only the design storage capacity is required as an input.

During the loading process, uranium decrepitates. Under some operating conditions, uranium sintering may also occur. Tailoring the operating conditions, fines production and transmission into the process loop may be reduced. A two year study to determine those conditions is underway.

During 1990, a special glass uranium bed will be constructed so that the dynamics of the uranium powder can be observed. In addition a carrier gas will be passed through the test bed. The fines which are entrained in this stream will be collected on an absolute filter. The size distribution of the particle which are collected is representative of the particle size distribution in the uranium bed. The change in this distribution with operating conditions will be monitored as part of a University of Toronto Masters program.

ZrCo offers many of the attractive features of uranium without the attendant disadvantages: radioactivity and pyrophoricity when exposed to air. A series of tests to determine the utility of ZrCo as a storage medium under conditions identical to uranium have been completed and indicate that with proper design ZrCo may offer a viable alternative as a storage medium.

During 1990, a new storage bed designed to eliminate the short comings identified in the previous year will be constructed and tested. Measurements using the original ZrCo bed will be used to generate "fast reacting" and "slow reacting" ZrCo alloy. Crystallographic studies are planned to determine if the change in the alloy structure induces a change in the reaction rate kinetics during hydriding.

Scavenger Beds

Scavenger beds are devices which preferentially scrub tritium from noble and nitrogen gas streams. Two types of scavenger beds are under development: those which can remove tritium from the effluent stream to the mCi/m^3 range and those which can purify the stream to the $\mu Ci/m^3$ range. Uranium is used in the former case while a Zr-Fe alloy is used in the latter case. Both types of beds are being developed for a West German nuclear facility. The latter type will also be used in the Research Division Tritium Laboratory as a part of the glovebox cleanup systems. The cross-sectional view of a typical uranium scavenger bed is shown in Figure 5. Gas enters along the central tube, flows to the bottom of the device and returns to exit by percolating through pie shaped cavities which contain uranium powder. In transit through the cavities, the tritium is stripped from the effluent stream. Detritiation factors exceeding 400 can be realized in a single pass. If the uranium is replaced with Zr-Fe alloy detritiation factors exceeding 30,000 have been measured.

A series of experiments are planned during 1990 to study the physics of multilayered uranium scavenger beds. These tests will be repeated using trace quantities of tritium to determine the performance of the devices at the ppb level.

Permeation

Experiments over the past few years have shown that under special conditions hydrogen can permeate through bimetallic layers at different rates depending on the permeation direction. The metal can be highly transparent to hydrogen in the forward direction and opaque to hydrogen in the reverse direction. This asymmetry in permeation has been exploited in several applications. Currently experiments are underway to utilize this feature in the purification of tokamak exhaust streams. The exhaust stream is passed through a low temperature plasma and in the process atomic tritium is separated by permeation through metal membranes from the main stream.

During 1990, factors which enhance the throughput through the bilayered membranes will be explored. Operating in the region of high throughput the impact of the impurities, water, methane, helium and nitrogen, in the hydrogen carrier stream will be studied.

Tritium Laboratory Development

Efforts in the Tritium Laboratory in 1989 centered on the commissioning of five gloveboxes. When completed, all the exhausts from each box will be plumbed directly to the main exhaust stack via separate tritium monitors. Nitrogen is the gas used in the boxes, and is delivered from a liquid nitrogen tank. Each box will be supported by its own glovebox cleanup system and an

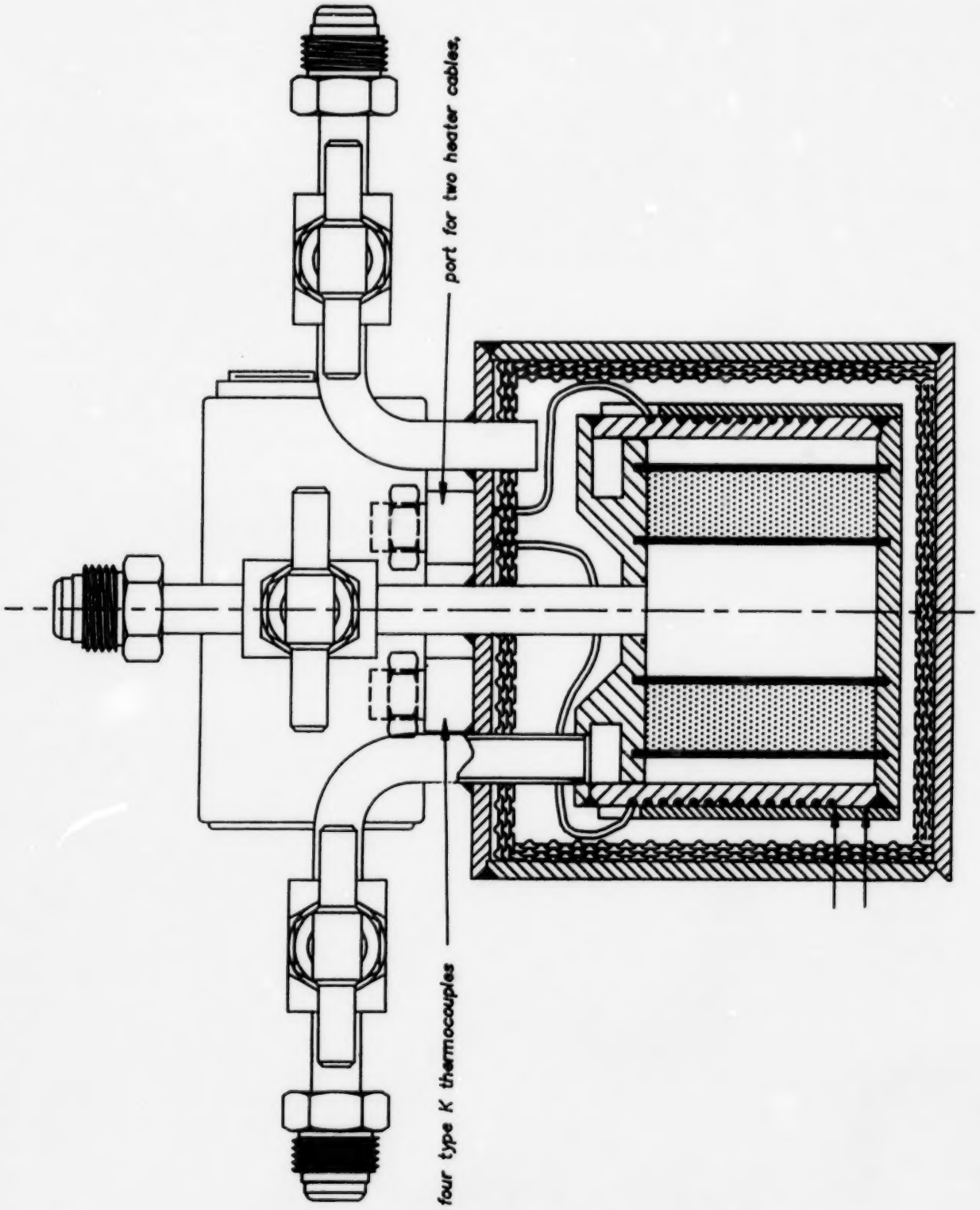


Figure 5

airlock air exchange system. Both of these systems are under development at Research and will be implemented in 1990. Several international facilities have expressed interest in the purchase of these systems.

Tritiated Light Development

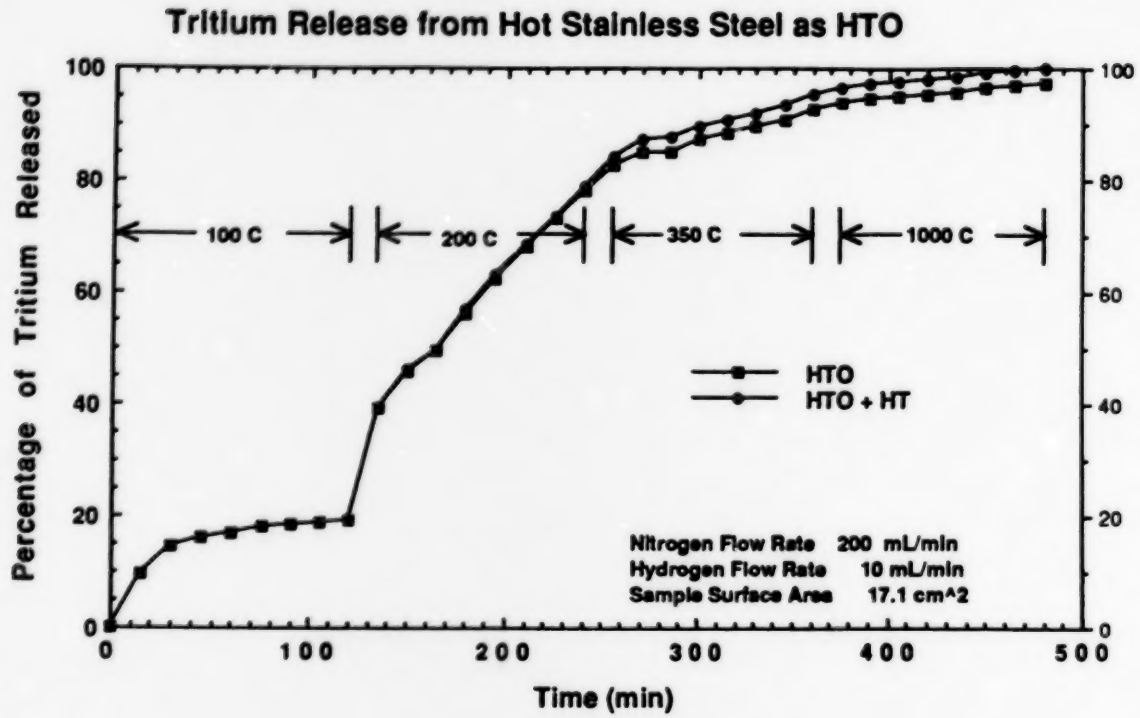
Gas tubes coated with a scintillant and filled with tritium have been used as a light source. These devices are being tested for airport runway lighting in remote locations and also in self powered emergency exit signs. However, they are vulnerable to breakage. An alternative scheme which uses tritium locked in thin titanium films and intimately mated with scintillant is being developed at Research. During 1989 three solid matrix tritium lights have been constructed. They are approximately one third as bright as gas lights.

During 1990, efforts will concentrate on increasing the brightness of these lights. Several avenues need to be explored. One of these is to tailor the film thickness to optimize the number of energetic electrons which reach the scintillant. Another, centers on increasing the mating area between the two media.

Decontamination of Stainless Steel

Part of the last cryogenic column at the Darlington Tritium Removal Facility (TRF) required replacement early in 1990. This operation required cutting and welding of tritiated stainless steel. The hazard associated with this operation was uncertain depending on the quantity of tritium bound to the surface, on the fraction which would be released during heatup, and on surface treatments, if any, prior to exposing the metal to air. To estimate the hazard level, samples with approximately similar surface contamination levels were heated in a nitrogen stream. The quantity and tritium species emitted as a function of temperature were measured. Typical results are illustrated in Figure 6. The majority of the tritium is released as tritiated water vapor. Only 20% of the bound tritium was released at 100°C while 90% was released at 350°C. Preconditioning the metal by heating the metal to 350°C for several hours was an effective way of reducing uncontrolled tritium emission.

Experiments will focus in 1990 on the physics of desorbing tritium from stainless steel surfaces. Tests to measure the bonding energy of the tritiated water on the metal surface are planned. The importance of the effluent stream in inducing water desorption will be measured.



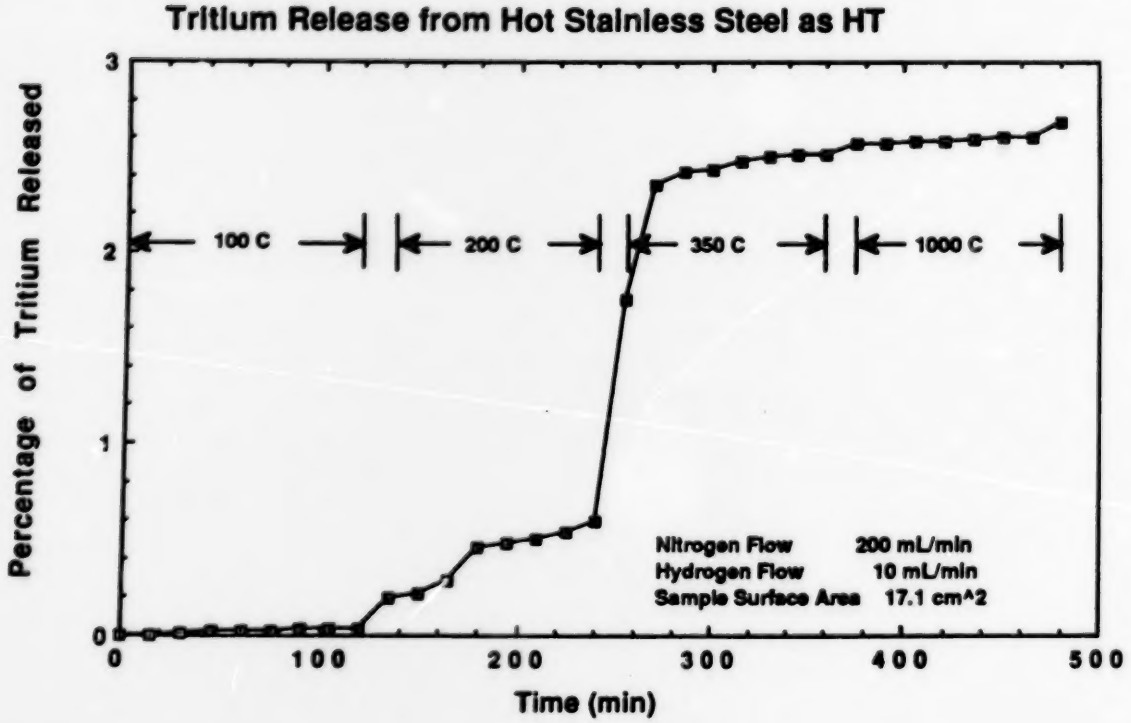


Figure 6 90-55

ACKNOWLEDGMENTS

The progress described in this report is the result of the dedicated and hard work of all members of the section. The work program is large and each individual has been involved in maintaining our longer term core research program, while at the same time contributing to many shorter term, but equally high priority projects. These individual contributions are gratefully acknowledged.

Approved by:

. Brown
Manager
Metallurgical Research Dept.

Prepared by:

P.C. Lichtenberger
Supervising Scientist
Corrosion & Tritium Technology Section
Metallurgical Research Department

PCL:km

TABLE 3

CORROSION AND TRITIUM TECHNOLOGY
STAFF AS OF YEAR END 1989

Dr. P.C. Lichtenberger	(Phil)	Supervising Scientist

a) *Corrosion Group*

Dr. B.D. Warr	(Brian)	Scientist
Dr. A.V. Manolescu	(Angela)	Engineer (on rotation)
Mr. P.J. King	(Peter)	Engineer
Mr. A.M. Brennenstuhl	(Alex)	Engineer
Dr. K.W. Lam	(Kim)	Engineer
Dr. G.I. Ogundele	(Gabriel)	Engineer (temporary)
Dr. M.B. Elmoselhi	(Mohammad)	Engineer (temporary)
Mr. A.C. Headon	(Al)	Senior Technologist
Mr. J.W. Mummenhoff	(Juergen)	Technician
Mr. E.M. Rasile	(Elvio)	Technician
Mr. S.W. Borda	(Steve)	Technician
Mr. L. Grant	(Laniel)	Technician
Ms. B. Guest	(Bonnie)	Technician
Mr. L. Chee	(Lloyd)	Technician
Mr. J. DeLuca	(Jerry)	Technician

b) *Tritium Technology Group*

Dr. W.T. Shmayda	(Walter)	Scientist
Mr. J.O. Noga	(John)	Engineer (on rotation)
Mr. A.G. Heics	(Andy)	Engineer
Mr. N.P. Kherani	(Nazir)	Engineer
Dr. A.B. Antoniazzi	(Armando)	Engineer (temporary)
Mr. K.D. Geiger	(Ken)	Technologist
Mr. B.L. Fishbein	(Bernie)	Technologist
Mr. M.J. Dudycz	(Mike)	Technologist
Mr. R.J. Opatril	(Rad)	Technician
Mr. M.D. Hooper	(Mark)	Technician

REPORTS ISSUED

DATE	CATEGORY	REPORT NO	JOB NO	FILE NO	TITLE AND AUTHOR(S)
April 11	B	M89-11-H	82-200-10122	827.42	Baseline Data for Water Heater Field Tests (A.C. Headon)
May 29	B	M89039-K		827.0	Corrosion & Tritium Technology Section 1988 Review and 1989 Work Program (P.C. Lichtenberger)
June 22	B	M89-45-K	82-875-300	827.0	NACE Annual Conference Corrosion/89 (G.I. Ogundele)
June 21	B	M89-30-P	82-211-40151	827.22	Preliminary Investigation of the Mechanism of Deuterium Uptake in Zirconium (M. Elmoselhi/B. Warr/C.A. Ward)
July 18	B	M89-65-K	82-213-40151	827.22	Sonthofen Trip Report (May 7-12, 1989) (M.B. Elmoselhi)
August 18	B	M89-38-K	82-142-50470	827.21	Examination of Reheater Tubes From Nanticoke Thermal Generating Station (NGS) (G. Ogundele)
November 9	B	M89-82-P	82-228-451	827.22	Electron Microscopical Analyses of Oxides in Zr-2.5 wt% Nb, (B. Warr/E. Rasile/A. Brennenstuhl)
November 17	B	M89-86-K	82-226-40162	827.24	H_2/D_2 Permeation Through Pd/23% at High Pressure, (A. Antoniazzi/W. Shmayda)
November 22	B	M89-61-[illegible]	82-231-451	827.22	Electron Microscopical Studies Zr-2.5 wt% Nb, (B. Warr/X. Meng/D. Northwood)
November 23	B	M89-100-K	82-249-462	827.28	Trip Report - IEEE 13th Symposium on Fusion Engineering, Knoxville, TN, October 2-5/89 (A. Heics)

REPORTS ISSUED

DATE	CATEGORY	REPORT NO	JOB NO	FILE NO	TITLE AND AUTHOR(S)
November 28	B	M89-101-K	82-250-400	827.28	Summary on the 36th American Vacuum Soceity Conference, Boston, MA, October 23-27/89, (A. Antoniazzi)
December 13	B	M89-105-K	82-875-100	827.4	NACE Canadian Region Eastern Conference, (A. Headon)
December 21	A	89-227-P	82-229-40151	827.22	Calculation of Deuterium Ingress Rates in Operating Reactor Pressure Tubes - Status Report (B. Warr/T. Byrne/D. Metzger)
December 11	A	89-167-P	82-231-451	827.22	OHRD Out-Reactor Autoclave Exposures Status Report - Role of Pressure Tube Material and Temperature on Corrosion and Hydrogen Uptake in Zr-2.5 wt% Nb, (B. Warr/J. Mummenhoff)

CONFERENCE/SEMINARS ATTENDED

VISITS TO OUTSIDE ORGANIZATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	ATTENDED BY	DESCRIPTION
April 12	P. King	Visit to CM^2, Ecole Polytechnique to discuss examination of steam generator tubing, Montreal, Quebec
April 17-21	G. Ogundele	NACE Corrosion 89, New Orleans
May 8	M. Elmoselhi	Third International Conference of Fundamentals of Adsorption - Sonthofen, W. Germany
May 22-26	W. Shmayda	Workshop on Cold Fusion, Santa Fe, New Mexico
May 13	A. Brennenstuhl	Parican, Seminar on Corrosion of High Alloy Steel Weldments, Toronto, Ontario
June 14 and 26	A. Brennenstuhl	Surface Science Western Analysis of I825 HX tube (removed from PNGS-B) and Analysis of Silicone Coated Samples, London, Ontario
June 19	G. Ogundele	COG Workshop on Erosion Corrosion, Toronto, Ontario
June 21	E. Rasile	TEM Specimen Preparation Course, Bethlehem, PA
June 26	W. Shmayda/N. Kherani	Canadian Association of Physicists (CAP) University of Guelph, Guelph, Ontario

CONFERENCE/SEMINARS ATTENDED

VISITS TO OUTSIDE ORGANIZATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	ATTENDED BY	DESCRIPTION
July 10	W.T. Shmayda	Metal Hydrides Conference, Tilton, N.H.
August 1	G.I. Ogundele	COG WP#19 Meeting, AECL-CRNL
Aug 5-9	A.M. Brennenstuhl/ P.J. King	Environmental Degradation of Materials in Nuclear Power Systems, Jekyll, Island, Georgia
Sept 1	W.T. Shmayda	KfK, Institut fur Radiochemie, Karlsruhe, West Germany
Sept 4	W.T. Shmayda	Max-Planck-Institut fur Plasmaphysik, Garching, W. Germany
Sept 6-14	W.T. Shmayda/N. Kherani	Tritium Technology: Int'l School of Plasma Physics Varenna, Italy
Sept 15-16	W.T. Shmayda/N. Kherani	Cold Fusion Workshop: Varenna, Italy
Sept 18	W.T. Shmayda/N. Kherani	Joint Research Center: Ispra, Italy
Sept 19	W.T. Shmayda/N. Kherani	Centre for Nuclear Research, Milan, Italy
Sept 21	W.T. Shmayda /N. Kherani	ENEA, CRE, Frascati, Italy
Sept 22	W.T. Shmayda/N. Kherani	ENEA, Casaccia, Italy
Sept 11	B. Warr	IAEA Conference, Portland, Oregon
Aug 16	P.C. Lichtenberger	Steam Generator Reliability Project Steering Committee, EPRI Offices, Palo Alto, CA

CONFERENCE/SEMINARS ATTENDED

VISITS TO OUTSIDE ORGANIZATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	ATTENDED BY	DESCRIPTION
October 2-5	A.G. Heics	IEEE 13th Symposium on Fusion Engineering, Knoxville, TN
October 23-27	A. Antoniazzi	36th American Vacuum Society Conference, Boston MA
October 28	G.I. Ogundele	Canadian Supply Institute, 1st Canadian Symposium on Taguchi Methods
November 11-18	A.M. Brennenstuhl	ESAT (Edinburgh) Evaluate LIMA, Edinburgh, Scotland
November 12-15	M.B. Elmoselhi/B. Warr	Fuel Channel Technology Seminar, Alliston, Ontario
November 12-15	P.C. Lichtenberger	Fuel Channel Technology Seminar, Alliston, Ontario
November 14	W.T. Shmayda	CRNL-T_2 Health and Safety Workshop, Chalk River, Ontario
December 8-17	P.C. Lichtenberger	USSR/Canada Fuel Channel Technical Exchange-Moscow/ Lenningrad
December 16	B. Warr/M. Elmoselhi	Vist to University of Western Ontario (I. Mitchell's Group)

VISITORS TO RESEARCH DIVISION

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	CONTACT	VISITOR AND ORGANIZATION	PURPOSE
April 17-18	P.J. King	Dr. P.M. Scott, UKAWA, Harwell	Presentation and discussion of Zr hydriding and nuclear plan corrosion
May 16-18	W. Shmayda/ N. Kherani	Dr. B. Cross, AECL	Paricipate in Cold Fusion Experiments
June	A. Headon	Kin, Dae Kyeong	Discussion of common interests re: Welding and materials R&D
June	A. Headon	G.I. Russell, Interprovincial Corrosion Control Co. Ltd.	
August 16	N. Kherani	Don Fielder/Peter Strong Brown & Rost Vickers Ltd. Defence & Industrial Contractors Hampshire, England	To learn of possible tritium technology support pertaining to an upcoming fusion related tender.
October 19	W. Shmayda/ N. Kherani	Prof. K. McNeill, J. Perz S. Zukotinski University of Toronto	To discuss getter research
October 20	W. Shmayda	Dr. L. Rosai: SAES, Milan Italy M. Ferris, SAES, Colorado Springs Colorado, U.S.A.	To discuss getter research
November	B. Warr	S. McIntyre, Universty of Western Ontario	To discuss surface analysis results
December 5	A. Headon	Mrs. Shu-te Kuo, Manager, Chemistry Lab Power Research Institute Taiwan Power Co.	Flyash corrosion experiments. Discussion of results and methods.

VISITORS TO RESEARCH DIVISION

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	CONTACT	VISITOR AND ORGANIZATION	PURPOSE
December 20	M. Elmosehi	Prof. Jan Spelt, University of Toronto	To discuss surface processes

PRESENTATIONS/PUBLICATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	AUTHORS	DESCRIPTION
June 19	W. Shmayda	Canadian Nuclear Society, Bruce NGS Search for Cold Fusion, Ontario
June 26	W. Shmayda, N. Kherani A. Antoniazzi	Looking for Cold Fusion, University of Guelph, Canadian Association of Physicists, Guelph, Ontario
August 16	P. Lichtenberger	Ontario Hydro Steam Generator R&D Program Overview EPRI Offices, Palo, Alto, CA
September 1	W. Shmayda	ZRCo Storage Bed Development at OHRD KfK: Institut fur Radiochemie, Karlsruhe, West Germany
September 4	W. Shmayda	T_2 Turbomolecular Pump Development, Max-Plank .nstitut für Plasmaphysik, Garching, West Germany
September 7	W. Shmayda	T_2 Storage Using Metal Hydrides: International School of Plasma Physics, Milan, Italy
September 8	W. Shmayda	T_2-Material Interactions-International School of Plasma Physics, Milan, Italy
September 11	B. Warr	IAEA Conference, Electron Microscopical Analysis of Zr-2.5 wt% Nb, Portland, Oregon
September 15	W. Shmayda/N. Kherani	Cold Fusion: A New Phenomenum or a Manifestation of Several Effects: IL Nuoro Gimento, Italian Physical Society
September 18	W. Shmayda/N. Kherani	T_2 Lab Activity at OHRD: Joint Research Center, ISPRA, Italy
September 21	W. Shmayda/N. Kherani	Status of Gas Driven Cold Fusion at OHRD: ENEA, Casaccia, Italy

PRESENTATIONS/PUBLICATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	AUTHORS	DESCRIPTION
September 25	A. Heics	CFFTP T_2 Safe Handling Course-T_2 Compatible Equipment, Toronto, Ontario
September 25	N. Kherani	Tritium-Materials Interactions-CFFTP Tritium Safe Handling Course, Toronto, Ontario
October 20	G. Longhurst/W. Shmayda	Uranium Storage and Accident Hazards Evaluation Fusion Technology, 1989
October 24	W. Shmayda	Bruce NGS; Cold Fusion Status
October 26	A. Antoniazzi	Presentation at 36th American Vacuum Society Conference RF Assisted Hydrogenic Pump, Boston, MA
October 26	N. Kherani	Cold Fusion, Canadian Nuclear Society, Ottawa, Ontario
November 13	P. Lichtenberger	An Overview of Hydrogen Ingress Mechanisms in Zr:Nb Pressure Tubes, Fuel Channel Technology Seminar, Alliston, Ontario
November 12-15	B. Warr/M. Elmoselhi E. Rasile/A. Brennenstuhl	Electron Microscopical Analyses of Oxides in Zr-2.5 wt% Nb, Fuel Channel Technology Seminar, Alliston, Ontario
November 12-15	B. Warr/A. Brennenstuhl N. McIntyre	Deuterium In-depth Analyses of Oxidized Zr-2.5 wt% Nb Pressure Tube Samples Removed from an Operating Nuclear Reactor, Fuel Channel Seminar, Alliston, Ontario
November 28	W. Shmayda	Current Research Trends in T_2 Technology, McMaster University, Hamilton, Ontario
November 29	M. Elmoselhi	Presentation of Preliminary Results of Measured D-Uptake by Zr-2.5 wt% Nb as a Function of Temperature at CRNL, Chalk River, Ontario

PRESENTATIONS/PUBLICATIONS

CORROSION AND TRITIUM TECHNOLOGY SECTION

DATE	AUTHORS	DESCRIPTION
November 23	P. Lichtenberger	Fuel Channel R&D Programs/Progress, Meeting with CPS Chemistry and Metallurgy Staff, Atrium
December 12	P. Lichtenberger	Current Understanding of Corrosion and Hydriding in Pressure Tubes, USSR/Canada Technical Exchange, Moscow
December 14	B. Warr/M. Elmoselhi	Review of OHRD PT Corrosion Research Program with CPS, Atrium